NUCLEAR FUELS FOR VERY HIGH TEMPERATURE APPLICATIONS

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Abstract

The success of the development of nuclear thermal propulsion devices and thermionic space nuclear power generation systems depends on the successful utilization of nuclear fuel materials at temperatures in the range 2000 to 3500 K. Problems associated with the utilization of uranium bearing fuel materials at these very high temperatures while maintaining them in the solid state for the required operating times are addressed. The critical issues addressed include evaporation, melting, reactor neutron spectrum, high temperature chemical stability, fabrication, fission induced swelling, fission product release, high temperature creep, thermal shock resistance, and fuel density, both mass and fissile atom. Candidate fuel materials for this temperature range are based on UO_2 or uranium carbides. Evaporation suppression, such as a sealed cladding, is required for either fuel base. Nuclear performance data needed for design are sparse for all candidate fuel forms in this temperature range, especially at the higher temperatures.

INTRODUCTION

The renewed interest in the development of space nuclear propulsion systems and high temperature thermionic-based space nuclear power systems brings forth a renewed interest in nuclear fuel forms that can be considered for these very demanding applications. The nuclear fuel that will generate the heat in these advanced concepts is probably the most critical and challenging component of these systems.

New, advanced space nuclear thermal propulsion (NTP) devices are being designed to produce large amounts of high performance rocket thrust using hydrogen heated in the reactor core to temperatures in excess of 3000 K.[1] Fuel temperatures in these designs will easily exceed 3200 K, and the heat generation rates in these fuels will be very high (typically >20 GW/m^3). Fuel burnup levels in these devices are usually anticipated to be considerably less than 1% FIMA. A key feature of all solid fuel forms proposed for use in NTP reactors is their high surface-to-volume ratios, which is dictated by the need to transfer the fission heat very rapidly to the propellant gas without overheating the fuel. High surface-to-volume ratio and high heat flux force the fuel structures or forms to be characteristically very thin and typified by geometries such as small spheres or thin plates. These extreme service demands and relatively fragile fuel structures will certainly limit the operating life of NTP reactors to only a few minutes or a few hours. Typical maximum NTP reactor operating times are not expected exceed 10 hours. Fuel forms based on uranium carbide are considered as prime candidates for the advanced NTP systems.

Advanced space nuclear power (SNP) systems that utilize thermionic devices for the generation of electricity are being designed with fuel element surface temperatures as high as 2250 K and peak fuel temperatures as high as 2600 K. The highest temperature thermionic SNP system concept, called the small ex-core heat pipe thermionic reactor (SEHPTR), involves the transfer of fission heat to the electrical generating devices by radiation across a vacuum gap.[2] Tungsten clad UO_2 is being considered for use in SEHPTR and in a slighdy older concept called the thermionic fuel element (TFE), which operates with fuel cladding surface temperatures closer to 1800 K.

The power densities of these reactors are typically much lower than for NTP reactors (typically <100 MW/m^3 for SNP reactors), but the operating times are much longer (up to 10 years). Fuel burnup levels in these devices are also anticipated to be low, typically less than 5% FIMA.

Some of the thermionic-based space nuclear power systems may also be used to generate satellite maneuvering thrust as well as electricity. A modest amount of rocket thrust might be produced by passing hydrogen through the same reactor core that produces heat for the generation of electricity.[3]

Very high temperature nuclear fuels have been used in only a few reactors over the years. The most notable of which are the test reactors built and operated in the NTP development program generally referred as Rover/NERVA. In one case, hydrogen propellant gas was heated to slightly over 2500 K in an experimental NTP reactor using fuel that consisted of pyrolytic-carbon-coated UC_2 particles imbedded in graphite structures that were coated with ZrC to minimize graphite attack by the hydrogen.[4] It is important to realize that even though these fuels performed reasonably well, they cannot be considered as fully developed even for the temperature regime that was tested in the Rover/NERVA program.

UO_2-tungsten cermets were considered very seriously for use in several NTP systems designed to heat hydrogen to temperatures similar to those produced in Rover/NERVA systems.[5-7] However, no reactors were ever built and operated that used this cermet fuel form.

This paper will discuss the fuel design issues for reactors that operate in the temperature range 2000 to 3500 K. We will also briefly indicate the state of the database for candidate fuels in this temperature range. However, we will limit our discussions to uranium bearing fuel forms that are based on uranium dioxide or the uranium carbides, and we will only consider those material-related design issues that we feel are representative of those that bear on system feasibility. We will not consider the uranium nitrides because of their lack of chemical stability in this temperature range.

HIGH TEMPERATURE NUCLEAR FUEL DESIGN ISSUES

Many of the design issues for high temperature nuclear fuels are unique and significantly different from those that are used in the design of reactors with fuel operating temperatures below 2000 K. The high-temperature nuclear fuel design issues are listed below in a rough order of decreasing importance.

Material evaporation...
Melting temperatures...
Uranium density...
Reactor neutron spectrum...
High temperature chemical stability...
Fabrication...
Fission product release...
Fuel swelling...
High-temperature creep...
Thermal shock resistance...
Mass density...

Fuel evaporation rates become very high at temperatures above 2000 K, and fuel evaporation is probably the most important limiting factor in the application of uranium-bearing nuclear fuels at high temperatures. This limitation is mostly governed by the high volatility of uranium which can be only partly suppressed by chemical combination to form more stable compounds such as UC_2 or UO_2.

Loss of material by high temperature evaporation is a time-temperature dependent process, so that probably the most significant set of design parameters is the evaporation loss rate as a function of temperature. A plot of evaporation loss rates of several candidate high-temperature fuels and tungsten is presented in Fig. 1 to indicate the magnitude of the evaporation problem at very high temperatures. The loss rates plotted in Fig. 1 are derived from data for evaporation into a vacuum except for the UC-ZrC* curve which represents the evaporation of $U_{0.05}Zr_{0.95}C_{1.07}$ into hydrogen gas maintained at 1 atmosphere pressure.[8] The evaporation rate of $U_{0.05}Zr_{0.95}C_{1.07}$ appears to be reduced from the vacuum evaporation rate by a factor of about eight.[8]

These loss rate curves imply that even the most refractory uranium-bearing materials will require some sort of containment to reduce material losses by evaporation to acceptable levels. A clear illustration of this point is seen by noting that the evaporation loss rate of the least volatile fuel form listed in Fig. 1, UC-ZrC*, is more than a micron per second at 3500 K even when suppressed by a hydrogen atmosphere.

Evaporation from these fuel forms also effects their chemical composition because of differences in the evaporation rates of the constituent elements. The equilibrium composition of a freely evaporating uranium carbides or oxides varies with temperature and is determined by the relative evaporation rates of the constituents. For example, the O/U atom ratio at 2300 K for UO_2 congruently evaporating in a vacuum is 1.94.[7] Most carbides are carbon deficient when congruently evaporating into a vacuum. Operating or processing environments may also alter the composition in relatively short times if there are chemical driving forces for reaction with one or more of the constituents.

The evaporation loss rate of tungsten is included in the plot in Fig. 1 because it is the least volatile of the refractory metals and it has been demonstrated as a compatible container material for UO_2 to very high temperatures. Tungsten has also been considered for a uranium carbide container material, but due to the fact that tungsten reacts with carbon-bearing materials to form W_2C at a significant rate,[9] the fuels with uranium carbides contained in tungsten may be limited to short operating times and temperatures below about 3060 K, the melting point of W_2C.

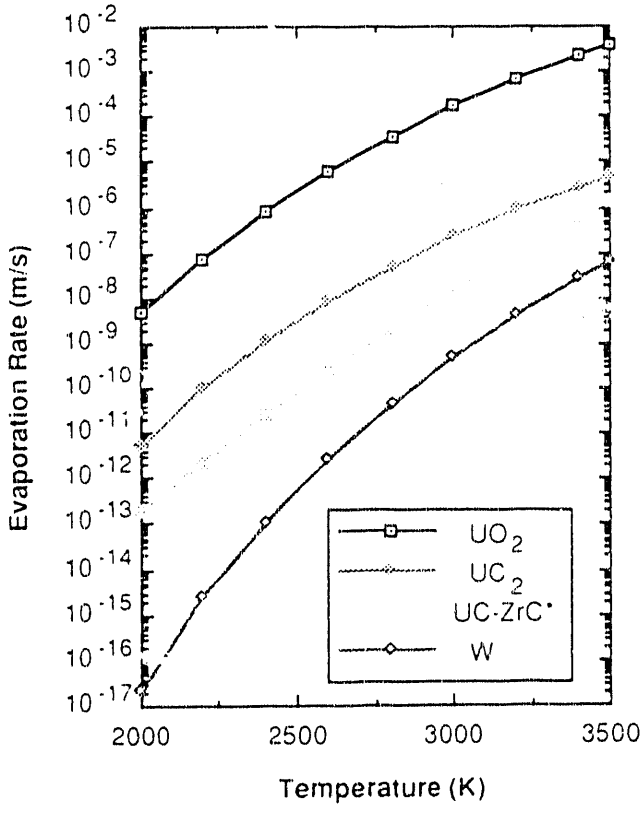

Fig. 1. Evaporation rates of several high temperature nuclear fuels and materials normalized to surface regression rates.

Melting represents the second most important design issue for high temperature nuclear fuels because it generally, but not necessarily, leads to loss of fuel structural integrity. Localized, contained melting of the fuel component might be

acceptable for short times in some cases. The melting temperatures of some high temperature fuel forms are listed in Table 1. As indicated in Table 1, uranium carbides mixed with refractory carbides, such as ZrC, are the only fuel forms that can be expected to remain solid at the upper end of the temperature range under consideration. However, as indicated by the large difference between the melting temperatures of $(U_{0.1}Zr_{0.9})C_{0.96}$ and $(U_{0.1}Zr_{0.9})C_{0.98}$, the melting temperatures of this type of material are very sensitive to the carbon content. The melting temperature of the W-60 v/o UO_2 cermet is based on the melting temperature of UO_2 even though the structural and geometric integrity of such a fuel form might be maintained for short times to much higher temperatures.

Table 1. Melting and uranium concentration data for candidate high temperature nuclear fuels.

Fuel Form	Melting Temperature (K)	U Density at 300 K (10^{28} atoms/m^3)
UC	2805[10]	1.6
UC_2	2835[10]	0.9
UC-40ZrC	3050[11]	1.0
$(U_{0.1}Zr_{0.9})C_{0.96}$	3550[12]	0.3
$(U_{0.1}Zr_{0.9})C_{0.98}$	3100[12]	0.3
W-60 v/o UO_2 Cermet	3075[10]	0.5

Uranium density is most important for fuels used in reactors whose neutron energy spectrum tends toward the high energy or fast flux regime. This is driven by the lower fission cross-sections for fast neutrons as compared to those for slow or thermal neutrons. The $(U_{0.1}Zr_{0.9})C_{0.96}$ fuel form listed in Table 1 has sufficient uranium density to be useable in a thermal reactor, but its uranium density is too low for use in most fast reactor designs.

The operating neutron spectrum of the reactor also has a major impact on the list of candidate materials that can be used in conjunction with the fuel material, such as cladding, cermet matrices or diluents. For example, significant quantities of natural tungsten are unacceptable in a thermal reactor because of the high resonance and absorption cross-sections for slower neutrons found for most of the naturally occurring isotopes, while it is practical for use in a fast reactor because few of those neutrons needed for fast fissioning of uranium are lost to interactions with the tungsten. Enrichment of natural tungsten in the ^{184}W isotope is possible and has been considered for use in thermal reactors.[13]

High temperature chemical stability of fuel forms is important in itself, but it is equally important for fuels to not be chemically attacked by the materials found in the operating environment. For example, hydrogen is the propellant of choice for NTP, but hydrogen can reduce or remove oxygen from UO_2 and cause the formation of liquid uranium in the temperature range under consideration. The importance of the maintenance of optimum carbon content in the carbides has already been indicated.

Fabrication is always an important design consideration for high temperature fuels, but it becomes a major issue for the complicated geometries required for extracting the high heat fluxes from nuclear thermal rocket engines. Most of the candidate high temperature fuel forms are difficult to fabricate into the desired delicate shapes primarily due to their brittle, refractory nature.

Fission product release is expected to be very high for all uranium-bearing fuels operating between 2000 and 3500 K. UO_2 fuel forms have been observed to release noncondensible fission products at their generation rates at 2000 K and above,[14,15] while the fission gas release from carbide fuels is considerably less at temperatures as high as 2580 K.[16] Fission product management is a major issue for both NTP and high temperature SNP applications.

Fission induced swelling behavior of fuels at high temperature is important to the operation of all NTP and high temperature SNP designs. Fuel swelling will generally lead to coolant passage closure in NTP reactor cores, while it will lead to mechanical distortion and loss of performance in thermionic SNP systems.

High temperature creep deformation in fuel forms must be maintained at low levels in both NTP and SNP cores, so that overall core distortions are kept to a minimum. In general creep, outside of that produced by fuel swelling and thermal stresses, is most significant to ground-test conditions where the forces of gravity can act on the reactor core. Creep failure of fuel claddings or coatings is generally unacceptable because it leads to problems such as uncontrolled fuel or fission product release.

Thermal shock resistance of high temperature fuel forms is essential to both NTP and SNP cores because thermal transients and gradients will be an integral part of the normal operating environment for any design and the candidate fuel forms tend to be brittle during reactor startup. The thermal shock resistance of a brittle solid material is proportional to its thermal conductivity and fracture strength, while it is inversely proportional to its elastic modulus and thermal expansion coefficient.

Because the high temperature fuel forms are to be used in devices that will be launched into space, it is important to consider their mass density. For example, carbide based fuel forms have mass densities considerably below those that include tungsten as a cladding of cermet matrix material.

OXIDE FUEL FORMS

In spite of relatively modest melting temperature and uranium density, UO_2 has potential for use in fuel forms operating to approximately 3000 K. The high volatility of UO_2 requires that this fuel form be contained inside some sort of sealed structure such as a cladding or a continuous metal matrix (cermet). There is some evidence that UO_2 can be successfully contained in tungsten near the UO_2 melting temperature for short times.[6,7,13] Molybdenum can be used as a containment material in the vicinity of 2000 K. Other refractory metals, such as niobium or tantalum, cannot be used for long-term, high-temperature containment of UO_2 due to chemical incompatibility. Times for successful containment are expected to be inversely proportional to the temperature as well as containment material dependent.

During fissioning at high temperatures, contained UO_2 fuel bodies quickly become hollow and strongly bonded to the inside of the container surfaces. Redistribution of UO_2 inside these containment structures occurs, primarily, by vapor transport to cooler surfaces. This process is very rapid, and in singly contained structures, it causes most of the void spaces

found in the virgin fuel to be collected and formed into a large cavity inside the fuel at locations furthest removed from the cooler regions.

The high vapor pressure of UO_2 demands that its containment be very reliable for high temperature applications. This requirement is difficult to achieve for long duration operation especially when the generated fission gases have to be contained or released in a controlled manner.

Under steady operating conditions, noncondensible, chemically inert fission products such as xenon or krypton are observed to be released at their generation rates throughout the entire temperature range after a brief incubation period.[14,15] Microstructural features, primarily interconnected grain boundary voids, developed in UO_2 during fissioning at high temperatures allow free passage of the fission gases to the outside.[17]

Very little is known about the behavior of UO_2 during irradiation at temperatures above 2000 K. Data on the irradiation swelling of UO_2 at the higher temperatures are sparse and somewhat questionable. However, an algorithm has been published in a material property library called MATPRO[18] that is claimed to be based on experimental data and able to predict the swelling of UO_2 to 2800 K. The swelling predicted by this algorithm is plotted as a function of both temperature and burnup in Fig. 2. Swelling data in the 2000 to 2200 K regime[14,15] fall close to the plotted surface within the burnup values indicated in the plot. A mechanistic computer code called FASTGRASS,[19] which calculates UO_2 swelling and fission product release, appears to underpredict swelling in the 2000 to 2200 K temperature range. Experimental data have not been found to compare with higher temperature swelling predictions which indicate reduced swelling with increased temperature.

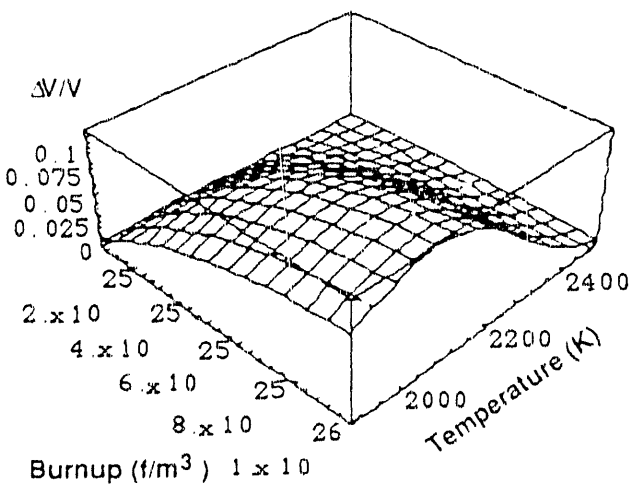

Fig. 2. UO_2 irradiation swelling predicted by the MATPRO algorithm.

Irradiation swelling data on the tungsten-UO_2 cermets[20] appear to follow the trends indicated in Fig. 2 over the temperature range 2000 to 2300 K and burnups in the neighborhood of 10^{26} f/m^3. However, the observed swelling is higher than predicted at the lower end of the temperature range, while near the upper end of the range the swelling is less than predicted by MATPRO.

Management of the released fission gases is necessary for safe and reliable operation, and fission gas venting without significant fuel loss or unacceptable redistribution within a reactor core is a major technological challenge to the utilization of UO_2 for long durations at these very high temperatures. Fuel cladding vent systems have been used in TFE designs to allow the fission gases to be released from the containment to prevent unacceptably high cladding stresses without significant loss of UO_2.[21,22]

Because UO_2 can be reduced in hydrogen at high temperatures, its chemical stability has been modified by the addition of more stable oxides such as Gd_2O_3[7] and ThO_2.[6] The incorporation of these mixed oxides into a tungsten clad tungsten cermet matrix further improves their chemical stability in hydrogen.

Thermal expansion mismatch between UO_2 and tungsten has been a concern especially for the application of the tungsten-UO_2 cermets to NTP. However, considerable testing has indicated that the tungsten matrix is sufficiently robust to withstand tens of thermal cycles to temperatures near 2800 K.[6,7] Good thermal shock resistance of the tungsten-UO_2 cermets was also demonstrated in these tests.

CARBIDE FUEL FORMS

As with UO_2, the evaporation rates from undiluted and uncontained uranium carbides are excessive above 2000 K. Also, as indicated in Table 1, their melting temperatures are not exceptional. Because UC_2 is typically used as a nuclear fuel along with carbon, the melting temperature listed for UC_2 in Table 1 is actually the UC_2+C eutectic temperature. However, as we have discussed, the evaporation of uranium carbides is effectively suppressed by dilution with less volatile, refractory carbides. Simple containment of the carbides has not been very successful for long durations for lack of a chemically compatible container material.

The melting temperature of UC can be effectively increased by forming solid solutions with higher melting metal carbides, such as ZrC, NbC, HfC, and TaC. In the (U,Zr)C system, for example, solidus temperatures in excess of 3570 K have been measured for compositions up to 10 mol% UC.[11] Increases in solidus temperature by adding NbC to (U,Zr)C have been measured.[11] Even higher solidus temperatures may be possible in solutions of UC with HfC and TaC.

The highest melting temperature compositions of the refractory metal carbides (ZrC, NbC, HfC, TaC) are hypostoichiometric in carbon. For example, the highest melting temperature in the Ta-C system, 4258 K, has a C/Ta ratio of 0.89.[10] However, both higher and lower carbon contents cause precipitous decreases in the solidus temperatures due to eutectics on the order of 300-800 K below the highest melting temperature. Eutectics with the metals are found in these systems at even lower temperatures than those with carbon, but the drop in solidus temperature with addition of metal (loss of carbon) from the highest melting composition is less steep. This limitation was recognized by fuel designers in the Rover/NERVA program.[12]

Measurement of the solidus at very high temperatures is difficult, and there is a need for confirmation of earlier measurements on the potential useful fuel systems (U,Zr,Nb)C and (U,Hf,Ta)C. Work in this area is underway.[23,24]

It will be difficult to maintain the optimal composition of the carbide fuels during high temperature operation in either an NTP or SNP reactor because of the chemical imbalances that will be imposed by the operating environment, including surrounding materials. For example, hydrogen will tend to react with all forms of uranium carbide to reduce their carbon contents and melting temperatures.

Similar to UC, (U,Zr)C fuels have exhibited large fission swelling rates at modest burnups. Ranken[16] has reported results on (U,Zr)C swelling, and these results are plotted in Fig. 3 along with results from UC. These results show that (a) the UC and (U,Zr)C data form one data set, (b) swelling is strongly temperature dependent, occurring at very large rates (>10%/10^{26} f/m^3) at temperatures above about 1870 K, and (c) swelling is negligible at fission densities less than $1x10^{25}$ f/m^3 (equivalent to ~0.3% FIMA) for theoretically dense $(U_{0.1}Zr_{0.9})C$ fuel. These data were obtained at fission rates considerably below that expected in NTP reactors. However, if fission rate does not significantly affect the swelling rate of the carbides, swelling is not likely to be a problem for NTP applications, but it remains a concern with (U,Zr)C fuels for long term applications where the burnup and temperature requirements combine to place the fuel in a region where swelling is large.

A reduction in volumetric swelling to 2.4% has been reported by Keller and Chubb[25] by the use of 20% porosity, combined with a 10 vol% centerline hole and 0.51 mm tungsten cladding with $(U_{0.1}Zr_{0.9})C$ irradiated in the temperature range 1870-2093 K to $1.9x10^{26}$ f/m^3 compared with 17% for similar $(U_{0.9}Zr_{0.1})C$ clad with 0.51 mm tungsten-26 wt% rhenium (W-26Re) and 40% for dense UC clad with 0.51 mm W-26Re. In this temperature range the tungsten cladding was enough stronger than the W-26Re cladding that the $(U_{0.9}Zr_{0.1})C$ fuel material filled the centerline hole first before causing external swelling. The fission gas retention in the 80% theoretical density $(U_{0.1}Zr_{0.9})C$ was 30% compared with 58% in dense UC.

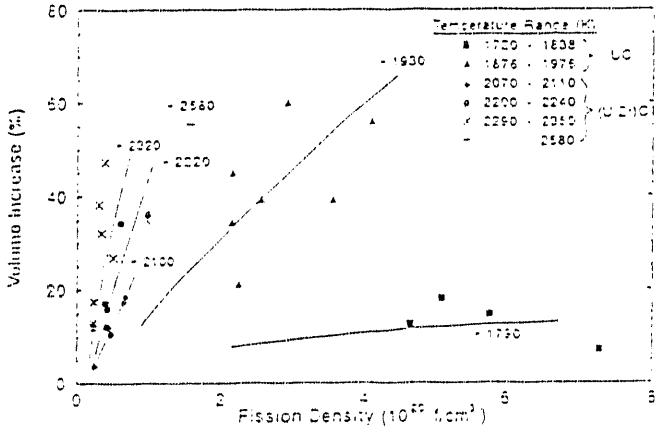

Fig. 3. Volumetric swelling of UC and (U,Zr)C as a function of fission density and irradiation temperature. (Reference 16)

Ranken[16] reported fission gas releases in the range 15-65% for (U,Zr)C containing 18 and 30% UC with porosities from 3 to 20%, and irradiated at temperatures from 2070 to 2580 K with burnups to $1.6x10^{26}$ f/m^3. He reported that his fission gas release data were not of sufficient quality to warrant correlation with temperature, porosity or burnup. Smith[26] measured the release of 66% of ^{85}Kr from $(U_{0.9}Zr_{0.1})C$ containing 24% as-fabricated porosity and small holes in the fuel body to enhance gas release and irradiated at a maximum temperature of 2035 K to a burnup of $3.0x10^{26}$ f/m^3(1.5% FIMA) and 85% release of ^{85}Kr from $(U_{0.5}Zr_{0.5})C$ containing 25% as-fabricated porosity and gas release holes and irradiated at a maximum temperature of 1995 K to a burnup of $2.1x10^{26}$ f/m^3(1.7% FIMA).

Smith[26] reported fractional retentions for other fission products. Releases derived from reported retentions at the locations corresponding to the maximum irradiation temperatures and closest to regions of cracks in the tungsten cladding were greater for the fuel containing 50% UC (1995 K maximum) than the fuel containing 90% UC (2035 K maximum). The values reported are ^{90}Sr (99 and 86%), ^{137}Cs (95 and 88%), ^{144}Ce (93 and 80%), and ^{95}Zr (19 and 15%). Kirk[27] reported fission product releases from fuels tested in the Nuclear Furnace 1 experiment. For the (U,Zr)C-graphite composite elements at the location of maximum fuel temperature (2500 K), the releases derived from reported retentions averaged 71% for ^{131}I, 64% for ^{89}Sr, 36% for ^{91}Y, and 30% for ^{140}Ba relative to ^{95}Zr. For the (U,Zr)C carbide elements at the location of maximum fuel temperature (2400 K), releases relative to ^{95}Zr were 15% for both ^{89}Sr and ^{91}Y and 13% for ^{140}Ba. These data indicate improved fission product retention for the carbide elements relative to the composite carbide-graphite elements.

The fission product releases derived from retention measurements of Smith[26] and Kirk[27] confirm expectations from studies on fission product retention in UC_2:[28] namely, fission gases and iodine will be mostly released, strontium and cerium carbides have relatively high vapor pressures and will be released to a large extent from UC and (U,Zr)C fuels, Cs remains metallic, is volatile and will be released from these fuels, and zirconium forms a stable carbide with a relatively low vapor pressure and will tend to be retained in the fuel to a greater extent than other fission products.

Fission product containment in the uranium carbide based fuel forms has traditionally been achieved by encapsulation or imbedding in chemically compatible matrix materials such as graphite (pure carbon) and overcoating with container materials, such as ZrC for high temperature applications, that can further inhibit fission product transport. These refractory carbides are also needed to protect the graphite matrices or coatings from attack by hydrogen in NTP applications.

Excessive mass losses from the cores of NTP reactors with uranium carbide based fuels due to evaporation and reaction with hydrogen has been determined to be a major concern. Both materials and environment control the losses. For example, NbC coated graphite fuel elements with pyrolytic carbon coated UC_2 were found to have unacceptably high loss rates from operating Rover/NERVA NTP test reactor cores due to the high carbon diffusion rate through the NbC coating. However, the use of ZrC coatings, which have lower carbon diffusion rates than NbC, did not totally solve the mass loss problem due to cracking of the coatings caused by irradiation effects to the substrate in low temperature regimes of the core. Excessive coating cracking was seen in those core regions where the thermal conductivity of the fuel element graphite matrix was seriously degraded by irradiation damage

causing a significant increase in the thermal strains, even under steady conditions.[12]

The thermal shock resistance of candidate uranium carbide based fuel forms is of considerable importance to fuel element design. Studies in the Rover/NERVA[12] program indicated that a graphite matrix composite containing 30 vol% $(U_{0.9}Zr_{0.1})C$ possessed the best thermal shock resistance of the fuels tested. Increasing concentrations of $(U_{0.9}Zr_{0.1})C$ caused a decrease in the thermal shock resistance. The graphite matrix provides those inherent properties previously listed that produce good thermal shock resistance. Monolithic carbide structures have much poorer thermal shock resistance than the composites because they have significantly lower thermal conductivities and higher elastic moduli than the matrix graphite. This comparative behavior was demonstrated in an NTP test reactor even though the thermal conductivity of the matrix graphite was seriously degraded in the low temperature region of the core, probably by fission fragment damage, which caused matrix surface cracks to develop.[12]

CONCLUSIONS

It has been concluded that evaporation is the most important fuel design issue for applications in the temperature range 2000 to 3500 K, and melting temperature has been concluded to the second most important fuel design issue. Other design issues have been defined that tend to be more closely related to the particular application and fuel material. However, there does not appear to be either sufficient experience or data bases to be assured that all of the critical design issues have been defined.

Fuel forms based on UO_2 and uranium carbide appear to be promising candidates for application in this temperature range, but data are sparse above 2000 K. Relevant, referencable data on the nuclear performance of UO_2 appears to extend to only about 2400 K, while similar data on uranium carbide based fuels extends only to about 2600 K. Experimental data directly relevant to reactor design and operation are virtually nonexistent near the upper end of the temperature range of interest. In short, the future development of advanced high temperature nuclear power or propulsion systems depends very heavily on the development of a full understanding of the fuel behavior in this severe environment.

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